UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

July 20, 2004

The Honorable Nils J. Diaz
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE WESTINGHOUSE ELECTRIC COMPANY APPLICATION FOR CERTIFICATION OF THE AP1000 PASSIVE PLANT DESIGN

Dear Chairman Diaz:

During the 514th meeting of the Advisory Committee on Reactor Safeguards, July 7-9, 2004, we completed our safety review of the Westinghouse Electric Company application for certification of its AP1000 passive plant design. This report is intended to fulfill the requirement of 10 CFR 52.53 that "the ACRS shall report on those portions of the application which concern safety." During our reviews, we had the benefit of discussions with representatives of Westinghouse, its consultants, and the NRC staff. We also had benefit of discussions with a member of the public and of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

The AP1000 design is robust and there is reasonable assurance that it can be built and operated without undue risk to the health and safety of the public.

DISCUSSION

<u>AP1000 Application</u>

On March 28, 2002, Westinghouse tendered its application to the NRC for certification of the AP1000 design. This application was submitted in accordance with Subpart B, "Standard Design Certification," of 10 CFR Part 52, "Early Site Permits; Standard Design Certification; and Combined License for Nuclear Power Plants," and Appendix O, "Standardization of Design: Staff Review of Standard Designs." The application was docketed on June 25, 2002 and assigned Docket No. 52-006.

The application consists of the AP1000 Design Control Document (DCD) and the AP1000 probabilistic risk assessment (PRA) report. The applicant originally submitted the AP1000 DCD on March 28, 2002. The DCD Tier 1 information contains inspection, tests, analyses and acceptance criteria (ITAAC), and Tirer 2 information describes design of the facility. Design certification is sought for the power generation complex, excluding those elements and features considered site specific. All safety-related structures, systems, and components (SSCs) are located on the nuclear island and are to be included in the design certification.

Three aspects of the plant design (instrumentation and control systems, human factors engineering, and some piping) will be completed by the combined license (COL) applicant using the design processes described in the DCD and ITAAC. A fourth aspect related to assurance of long-term recirculation cooling following a LOCA will be confirmed by the COL applicant using NRC guidance that is approved at that time.

The staff issued a draft safety evaluation report (DSER) on June 16, 2003, and an advanced copy of final safety evaluation report on May 25, 2004.

AP1000 Design Description

The AP1000 design is similar in concept to the AP600 design, but provides much higher power levels. To accommodate the higher power [1000 Mwe for AP1000 compared to 600 Mwe for AP600], the following systems and components were increased in size and/or capacity for AP1000 over those of AP600:

- Core length and number of assemblies
- Key NSSS components
  - height of reactor vessel
  - steam generators
  - canned motor reactor coolant pumps
  - pressurizer
- Containment height (volume)
- Capacities of passive safety system components
- Automatic depressurization system (ADS) stage-4 squib valve
- Turbine capacity

As was the case for AP600, the AP1000 design is intended to meet the safety requirements and goals defined for advanced light water reactors with passive safety features specified in the Electric Power Research Institute Utility Requirements document. The plant consists of five principal structures: the nuclear island, the turbine building, the annex building, the diesel generator building, and the radwaste building. The nuclear island includes all safety-related and seismic Category 1 structures and is designed to withstand the effects of natural phenomena and postulated events. It consists of a containment building, a concrete shield building, and an auxiliary building.

The containment building consists of a free-standing, 1¾ inch thick steel containment vessel which has a total free volume of about 2 million cubic feet and a design pressure of 59 psig. The vessel performs the function of limiting the release of radioactivity to the atmosphere for postulated design basis accidents and is part of the passive containment cooling system. In the event of an accident, the passive containment cooling system releases water which runs down the outside surface of the containment vessel to enhance heat removal.

The shield building comprises the structure and annulus area that surrounds the containment vessel. The annulus is configured to complete the passive containment cooling function by providing for natural convection of the outside air up along the containment vessel and out the top.

The auxiliary building is designed to provide protection and separation for the seismic Category 1 mechanical and electrical equipment located outside the containment building. The building also provides protection for safety-related equipment against the consequences of internal and external events. The main control room, Class 1E I&C systems, Class 1E electrical systems, and reactor fuel handling areas are contained in the auxiliary building.

The turbine building houses the main turbine generators and associated fluid and electrical systems. The annex building includes the health physics area, the technical support center, access control, and personnel facilities. The diesel generator building houses two diesel generators and their associated support systems. The radwaste building contains facilities for handling, processing, and storing radioactive waste.

The overall plant arrangement utilizes building configuration and structural designs to minimize the building volumes and quantities of bulk materials (concrete, structural steel, rebar) consistent with design basis safety, operational, maintenance, and structural needs. The plant arrangement provides separation (generally by concrete walls) between safety and non-safety equipment to preclude adverse interactions among them. Separation between redundant safety equipment provides confidence that the safety functions can be performed.

The ITAAC program is intended to ensure that the plant, when built, will conform to the design parameters and assumptions that existed at the time of design certification.

Safety Enhancement Features

The AP1000 design has enhanced safety features similar to those in the AP600 design. These include an improved reactor core design, a large reactor vessel, a large pressurizer, an in-containment refueling water storage tank (IRWST), an automatic depressurization system, a digital microprocessor-based I&C system, hermetically sealed canned motor coolant pumps mounted to the steam generators, and increased battery capacity. The AP1000 design has a defense-in-depth provision for external flooding of the reactor vessel which is intended to provide for in-vessel retention of any accident-induced core melt. The reactor vessel has no bottom head penetrations. This both reduces the potential for a LOCA that would quickly drain the vessel and enhances natural circulation cooling via the cavity flood water.

The AP1000 safety approach is to credit only passive systems to meet all the design basis accident (DBA) requirements with only a one time realignment of valves. Available active pumps, diesels, AC power, cooling water, HVAC, I&C, etc., are not required. The active non-safety-related systems support normal operation and minimize challenges to the passive safety systems. Although these systems are not credited in the safety evaluation case, they provide additional defense-in-depth. The regulatory treatment of nonsafety systems (RTNSS) process was used to impose special requirements on some nonsafety systems to ensure, with high confidence, that they would be available when needed. These systems provide redundancy and diversity that contribute to achieving very low values for core damage frequency (CDF) and large release frequency (LRF).

Probabilistic Risk Assessment

The AP1000 design certification application included a PRA in accordance with regulatory requirements. This PRA was done well and rigorous methods were used. We found that this PRA was acceptable for certification purposes. The mean estimates of the risk metrics are:

CDF:.......... 5 E-07/yr
LRF:........... 6 E-08/yr

These risk metrics are well within the agency's expectations for advanced plants. The fact that the PRA was an integral part of the design process was significant to achieving this estimated low risk.

ACRS Review Approach

The ACRS review activities for the AP1000 design certification are listed in the appendix to this report. These activities should be viewed in concert with all the ACRS review activities conducted for certification of the AP600 design. The AP1000 design is similar in concept to the AP600 design. Consequently, the ACRS's approach to reviewing the certification application for the AP1000 design was to become familiar with the changes from the AP600 design made to accommodate the increased power level compared to AP600 and to assure ourselves that these did not pose any new safety considerations or result in an unacceptable increase in risk. As part of this approach, the new phenomena identification and ranking table (PIRT) was reviewed to see if any new phenomena were identified and if there were any significant changes in rankings of events and phenomena identified for AP600. The new AP1000 scaling analyses were also reviewed to determine what portions of the AP600 test and analysis program were directly applicable to the AP1000 design.

We concluded that most of the previous AP600 review findings were applicable to the AP1000 design. This conclusion greatly enhanced the efficiency of the reviews of the AP1000 safety assessments. We had a number of subcommittee and full- Committee meetings to review the AP1000 as listed in the Appendix. Our reviews did not address security related issues.

During these reviews, we issued three letters identifying issues of concern and areas for which we needed additional discussions. We agreed with the staff's proposed resolutions [Reference 8] of all but two of these issues. In the case of the "in-vessel retention" and "organic iodine production" issues, we developed our own arguments for the resolution. Thus, all ACRS issues have been resolved.

We also discussed the concerns expressed by a member of the public. Most of these are "process" related and are within the purview of the staff. We considered one technical item, the effect of solar heating on the passive containment cooling system's ability to deal with design basis accidents. Westinghouse has assumed a conservative containment cooling water temperature of 120 °F and an air inlet temperature of 115 °F. They have also proposed a 120 °F technical specification limit on the containment cooling water temperature. We find these values to be sufficiently conservative for design basis evaluation.

The ACRS reviewed the NRC staff's safety evaluation and concluded that the staff had done a comprehensive and competent assessment of the compliance of the AP1000 design with the required regulations.

Sincerely,

/RA/

Mario V. Bonaca
Chairman

References: See next page

References:

1. U.S. Nuclear Regulatory Commission, "Draft Safety Evaluation Report Related to Certification of the AP1000 Standard Design" Volumes 1 and 2, dated June 16, 2003.

2. U.S. Nuclear Regulatory Commission, "Advanced Copy of the Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design" Volumes 1 and 2, dated May 25, 2004 (Predecisional Information).

3. Westinghouse Electric Company, AP1000 Design Control Document (DCD), APP-GW-GL-700, Revision 11, dated May 20, 2004.

4. Report dated July 23, 1998, from R. L. Seale, Chairman, ACRS, to Shirley Ann Jackson, Chairman, NRC, Subject: The Safety Aspects of the Westinghouse Electric Company Application for Certification of the AP600 Passive Plant Design.

5. Letter dated June 21, 2000, from J. T. Larkins, Executive Director, ACRS, to W. D. Travers, Executive Director for Operations, NRC, Subject: AP1000 Pre-Application Review.

6. Report dated March 14, 2002, from G. E. Apostolakis, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Phase 2 Pre-Application Review for AP1000 Passive Plant Design.

7. Letter dated March 17, 2004, from M. V. Bonaca, Chairman, ACRS, to W. D. Travers, Executive Director for Operations, NRC, Subject: ACRS Reviews of the Westinghouse Electric Company Application for Certification of the AP1000 Plant Design-Interim Letter.

8. Letter dated May 18, 2004, from W. D. Travers, Executive Director for Operations, NRC, to M. V. Bonaca, Chairman, ACRS, Subject: Response to ACRS Interim Letter Regarding the ACRS Reviews of the Westinghouse Electric Company Application for Certification of the AP1000 Plant Design.

9. Letter dated April 30, 2004, from R. P. Vijuk, Manager, Westinghouse Electric Company, to U. S. Nuclear Regulatory Commission, Subject: Westinghouse Responses to ACRS Open Items.

10. Letter dated May 11, 2004, from R. P. Vijuk, Manager, Westinghouse Electric Company, to U.S. Nuclear Regulatory Commission, Subject: Transmittal of Revised Responses to AP1000 DSER Open Items.

11. Letter dated July 30, 2003, from S. G. Sterrett, Assistant Professor, Duke University, to ACRS Subcommittee on Future Plant Designs, Subject: AP1000 Fluid Systems Design & QA Procedures.

12. Letter dated July 31, 2003, from S. G. Sterrett, Assistant Professor, Duke University, to ACRS Subcommittee on Future Plant Designs, Subject: Heat of Solar Radiation and AP1000 Ultimate Heat Sink.

13. Letter dated April 20, 2004, from J. E. Lyons, Program Director, NRC, to S. G. Sterrett , Assistant Professor, Duke University, Subject: Response to Concerns About The AP1000 Design Certification.

14. Letter dated July 8, 2004, from S. G. Sterrett, Assistant Professor, Duke University, to ACRS Members, J. P. Segala (AP1000 Project Manager, NRC), and J. E. Lyons (Program Director, NRC), Subject: NRC Response to Concerns About AP1000 Design Certification.

APPENDIX

CHRONOLOGY OF THE ACRS REVIEW OF THE WESTINGHOUSE APPLICATION FOR THE AP1000 PASSIVE PLANT DESIGN CERTIFICATION

The extensive ACRS review of the AP1000 design and its interactions with representatives of the NRC staff and Westinghouse are discussed in the minutes and transcripts of the following ACRS meetings.

ACRS MEETING/DATES	SUBJECT
475th ACRS Meeting 8/29-9/1/2000	Issues identified during AP1000 pre-application Review (Phase 1)
Thermal-Hydraulic Phenomena 3/15/2001	Westinghouse proposed approach to address AP1000 thermal-hydraulic issues
481th ACRS Meeting 4/5-7/2001	Thermal-Hydraulic issues associated with the AP1000 passive plant design
Thermal-Hydraulic Phenomena 2/13-14/2002	Phase-2 pre-application review; application of AP600 test programs and analyses codes to AP1000 design
Future Plant Designs 2/14-15/2002	Phase-2 pre-application review; use of design acceptance criteria (DAC), and regulatory exemptions for AP1000 design
490th ACRS Meeting 3/7-9/2002	Phase-2 pre-application reviews of the AP1000 Design
497th ACRS Meeting 11/7-9/2002	Westinghouse AP1000 design-review schedule
Reliability and Probabilistic Risk 1/23-24/2003	Reliability of AP1000 design components; ADS-4 squib valve function
499th ACRS Meeting 2/6-8/2003	PRA Subcommittee Chairman report on the PRA for AP1000 design
Thermal-Hydraulic Phenomena 3/19-20/2003	Review of liquid entrainment issue for AP1000 design
501st ACRS Meeting 4/10-12/2003	Thermal-hydraulic Subcommittee Chairman report on thermal-hydraulic issues for AP1000 design
Thermal-Hydraulic Phenomena 7/16-17/2003	Review of sump strainer, boron precipitation, ADS-4 squib valve, and computer codes

Future Plant Designs 7/17-18/2003	**Review I&C, man-machine interface, materials, leak-before-break, human factors, and DSER open items**
506th ACRS Meeting 10/1-4/2003	**Interim review of the AP1000 design, and resolution of open items**
Thermal-Hydraulic Phenomena 2/10-11/2004	**Thermal-hydraulic issues for AP1000 design, computer codes, and core-level during long-term cooling**
510th ACRS Meeting 3/3-6/2004	**Interim review of AP1000 design, open items and ACRS concerns**
513th ACRS Meeting 6/2-4/2004	**NRC staff's response to the ACRS report of March 17, 2004 for the AP1000 design**
Future Plant Designs 6/25/2004	**Review of NRC staff's response to ACRS concerns, technical issues, any remaining open items, and FSER review**
514th ACRS Meeting 7/7-9/2004	**Final ACRS review of the AP1000 design**